The Enduring Legacy of ACRS: Contributing to Safety-Licensing Review of Reactor Facilities

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ABSTRACT

Since 1957, the Advisory Committee on Reactor Safeguards (ACRS) has had a continuing statutory responsibility for providing independent reviews of, and advising on, the safety of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards in the United States. This white paper begins with a history of the ACRS, noting some of its significant contributions to reactor safety. The paper then presents a historical perspective on ACRS reactor licensing reviews. The essential role of the ACRS on reviewing the currently proposed new reactor designs, which are radically different from the current fleet of light water reactors, is also discussed.

The views expressed in this report are solely those of the author and do not necessarily represent the views of the ACRS.

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ABBREVIATIONS

ABB-CE	ASEA Brown Boveri - Combustion Engineering
ABWR	Advanced Boiling Water Reactor
ACNW	Advisory Committee on Nuclear Waste
ACNW&M	Advisory Committee on Nuclear Waste and Materials
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
AP600	Advanced Passive 600
AP1000	Advanced Passive 1000
ATWS	Anticipated Transients Without Scram
BWR	boiling water reactor
CP	construction permit
CRBRP	Clinch River Breeder Reactor Plant
DCA	design certification application
DI&C	digital instrumentation and control
DOE	Department of Energy
ECCS	emergency core cooling system
ESBWR	Economic Simplified Boiling-Water Reactor
FACA	Federal Advisory Committee Act
FFTF	Fast Flux Test Facility
GCFBR	gas cooled fast breeder reactor
GDC	General Design Criteria
GE	General Electric
GENE	General Electric Nuclear Energy
GSI	Generic Safety Issue
HTGR	high temperature gas-cooled reactor
LMR	liquid metal reactor
LOCA	loss-of-coolant accident
LWR	light water reactor
MHTGR	modular high temperature gas cooled reactor
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
NSRRC	Nuclear Safety Research Review Committee

PCCS	passive containment cooling system
PCRV	prestressed concrete reactor vessel
PHWR	pressurized heavy-water reactor
PRA	probabilistic risk assessment
PRDC	Power Reactor Development Company
PRISM	Power Reactor Innovative Small Module
PSER	preapplication safety evaluation report
PWR	pressurized water reactor
RG	regulatory guide
SAFR	Sodium Advanced Fast Reactor
SE	safety evaluation
US-APWR	U.S. Advanced Pressurized-Water Reactor

1. INTRODUCTION

The 1957 amendment to the Atomic Energy Act of 1954 established the Advisory Committee on Reactor Safeguards (ACRS) as a statutory committee with its independent advisory role, and the responsibility to "review safety studies and facility license applications..." and to advise the U.S. Atomic Energy Commission (AEC) "with regard to the hazards of proposed or existing reactor facilities and the adequacy of reactor safety standards."

With the enactment of the Energy Reorganization Act of 1974, the ACRS was assigned to the newly established NRC with its statutory requirements intact.

The ACRS consists of up to 15 members who are well recognized experts in technical areas that are key to nuclear safety and with a breadth of experience in all aspects of the nuclear enterprise: industry, universities. national laboratories, and government.

The present-day matters referred to the Committee can be grouped into five categories: licensing reviews, regulatory policies and practices, operating reactors safety oversight, safety research reviews, and nuclear materials and waste. The scope of the present day ACRS review activities include:

- Specific statutory review functions established in NRC regulations

- Reviews of any generic issues or other matters referred to it by the Commission for advice

- Reviews of specific generic matters or nuclear facility safety-related items, on ACRS' own initiative (per 10 CFR 1.13)

- Upon request from the Department of Energy (DOE) and with the consent of the Commission, ACRS provides advice on U.S. naval reactor designs, and hazards associated with DOE nuclear activities and facilities.

- Upon request and with the consent of the Commission, ACRS provides technical advice to the Defense Nuclear Facilities Safety Board (per the National Defense Authorization Act of 1989).

ACRS operations are governed by the Federal Advisory Committee Act (FACA), which is implemented through NRC regulations (10 CFR Part 7). ACRS operational practices encourage the public, industry, State and local governments, and other stakeholders to express their views on matters related to safety of reactor facilities. All ACRS records, reports, transcripts, or other documents, which are made available to or prepared for or by the Committee, are publicly available subject to the provisions of the Freedom of Information Act (5 U.S.C. 552) and NRC's Freedom of Information Act regulations at 10 CFR part 9, sub-part A.

Throughout its history, the ACRS review has been an important element of the reactor licensing process. The Committee's licensing reviews led to evolution of many new safety requirements and design changes dealing with a wide range of technical issues.

As NRC is preparing for review of new reactor designs that are radically different from the current fleet of LWRs, the role of ACRS, with its diverse technical expertise, will continue to be

essential for integrated/multi-disciplinary independent review and advice.

This white paper begins with a brief history of the ACRS, noting some of its significant contributions to reactor safety. It then presents a historical perspective on ACRS reactor licensing reviews. The essential role of the ACRS on reviewing the new advanced non-light-water reactor designs is also discussed.

2. A BRIEF HISTORY OF ACRS

2.1 Creation of ACRS

The history of ACRS goes back to 1947 when AEC soon after its establishment recognized the need for an independent technical group to review and provide advice on reactor safety matters and thus a Reactor Safeguard Committee, chaired by Dr. Edward Teller, was established. Dr. Teller has been quoted saying that the Reactor Safeguard Committee "was about as popular - and as necessary - as a traffic cop" [Mazuzan & Walker, 1985]. As stated by Richard Meserve, former NRC Chairman, the Reactor Safeguard Committee "clearly established an enduring characteristic of the ACRS – a willingness to provide candid views on reactor safety issues, even at the risk of taking unpopular positions" [NRC, 2003].

Dr. Edward Teller
Chairman of Reactor Safeguard Committee, 1947-1953

In 1950, the AEC established a second advisory committee, the Industrial Committee on Reactor Location Problems, charged with the responsibility of advising on what we would today consider siting issues, including seismic and hydrological characteristics of proposed sites. In 1953, the Reactor Safeguards Committee and the Industrial Committee on Reactor Location Problems were combined by the AEC and the ACRS was formally born.

Dr. C. Rogers McCullough
First ACRS Chairman, 1953-1960

2.2 ACRS as a Statutory Committee

In 1957, an amendment to the Atomic Energy Act of 1954 established the ACRS as a statutory committee advising the AEC. According to Section 29 of the Act the "Committee shall review safety studies and facility license applications referred to it and shall make reports thereon, shall advise the Commission with regard to the hazards of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards and shall perform other such duties as the Commission may request." Subsection 182b of the Act requires that the Committee's report "be made part of the record of the application and available to the public, except to the extent that security classification prevents disclosure."

Establishing the ACRS as a statutory committee resulted, in part, from a controversy involving licensing of the Fermi-1 liquid metal fast breeder reactor.

An account of this, as reported by J. Samuel Walker and Thomas R. Wellock [Walker & Wellock, 2010], is summarized here.

In 1956, the Power Reactor Development Company (PRDC), a consortium of utilities led by Detroit Edison Company, applied for a construction permit (CP) to build a fast breeder reactor, located on Lake Erie within 30 miles of both Detroit and Toledo. The fast breeder reactor that PRDC planned was far more advanced in its technological complexity than the light-water designs proposed in earlier applications. After review of PRDC's application and discussions with company representatives, the ACRS concluded in an internal report to the Commission that there was insufficient information available at the time to give assurance that the PRDC reactor could be operated without public hazard. ACRS also expressed doubt that its safety concerns could be resolved within PRDC's proposed schedule for obtaining an operating license. The ACRS urged that the AEC expand its experimental programs with fast breeder reactors to seek more complete data on the issues raised in the PRDC application.

During congressional hearings, members of the Joint Committee on Atomic Energy (the AEC congressional oversight committee) were troubled by revelations of safety concerns and the AEC Chairman's intention to attend the groundbreaking ceremony for a reactor whose construction permit was still being evaluated by the AEC. They were particularly disturbed by the AEC's failure to inform them about the ACRS reservations. The AEC was obligated by the Atomic Energy Act of 1954 to keep the Joint Committee "fully and currently informed" about its activities, and Joint Committee members believed that, in the case of the ACRS report, the agency had failed to carry out its responsibility.

The AEC was unwilling to provide a copy of the ACRS report to the Joint Committee without the condition that it would be kept "administratively confidential." The AEC also refused to provide a copy of the ACRS report to the State of Michigan on the grounds that "it would be inappropriate to disclose the contents of internal documents." Meanwhile, the AEC was completing its review of PRDC's application. The AEC took a more optimistic view of the safety of the proposed reactor than ACRS had. Since the company had agreed to perform tests to answer the questions raised by ACRS, the AEC decided to issue the construction permit. However, it acknowledged the ACRS concerns by inserting the word "conditional" in the construction permit to emphasize that the company would have to resolve the uncertainties about safety before it could receive an operating license.

Fermi Unit 1: the world's first commercial liquid-metal fast breeder reactor. Establishing the ACRS as a statutory committee resulted, in part, from a controversy involving licensing. Fermi 1 started operation in 1963.In October 1966, it experienced fuel melting from partial core flow blockage. Three years and nine months later, with cleanup completed and fuel replaced, Fermi 1 was restarted. In November 1972, the PRDC made the decision to decommission Fermi 1 for economic reasons.

To prevent a recurrence of the AEC's conduct in the PRDC case, the Joint Committee soon introduced legislation to establish ACRS as a statutory body, direct that its reports on licensing cases

be made public and require public hearings on all reactor applications. The AEC opposed all three measures but muted its objections because they were presented as amendments to a bill to provide insurance coverage for reactor owners, which the agency strongly favored. Establishment of the ACRS as a statutory committee was accompanied by a significant expansion of public access into the regulatory and licensing activities of the AEC.

2.3 Role of ACRS Over Its History

The role of ACRS has evolved over its history. The early licensing reviews were generally based on the engineering experience and judgment of regulatory staff working closely with the ACRS, without the availability of the regulatory guidance and structure established later during commercial development of LWRs. Most of today's U.S. nuclear power plants were licensed during the 1960s and 1970s, when both the technology and its governing regulations were in the formative stages. The Committee's licensing reviews led to evolution of many new safety requirements and design changes dealing with a wide range of technical issues [Nourbakhsh, 2018a]. The ACRS reactor licensing reviews are further discussed in Section 3 of this white paper.

Early in the development of commercial nuclear power, the ACRS became concerned with core meltdown accidents, particularly one in which the plant's emergency core cooling system (ECCS) might fail to operate as designed, could lead to a breach of containment. In 1966, at the "prodding" of ACRS, the AEC established a special task force to look into the problem of core meltdown [Walker & Wellock, 2010]. The task force, chaired by William K. Ergen, a former ACRS member, issued its report in October 1967 [Ergen, 1967]. The report offered assurances about the reliability of ECCS designs and improbability of a core meltdown, but it also acknowledged that a loss-of-coolant accident (LOCA) could cause a breach of containment if the ECCS failed to perform. In an ACRS letter on the task force report, dated February 26, 1968, the Committee strongly recommended that a "positive approach be adopted toward studying the workability of protective measures to cope with core meltdown" [ACRS, 1968a]. The Committee also recommended, as it did in its 1966 report on safety research, that a "vigorous program be aimed at gaining better understanding of the phenomena and mechanisms important to the course of large-scale core meltdown." The task force report and ACRS recommendations formed the basis of some of the most important research initiatives and regulatory decisions by the AEC and the NRC, including the AEC's decision to undertake a study to estimate the probability of a severe accident, which resulted in the publication of the landmark Reactor Safety Study (WASH-1400) [NRC,1975] and the beginning of the science of probabilistic risk assessment as applied to nuclear power plant safety.

As the ACRS moved into the 1980s, and continuing to the present day, the Committee shifted much of its attention from plant design and construction to improvements in both the operation and regulation of nuclear power plants [Nourbakhsh, 2018b]. The ACRS has made valuable contributions over a wide range of issues at operating plants, including fire safety, operator training and human performance, digital instrumentation and control (DI&C) upgrades, extended power uprates, plant aging, and license renewal.

A list of generic items related to construction or operation of light-water reactors, which was developed initially by

the ACRS, and the work of the NRC staff to resolve those items became steadily more formal, stemming from the requirement of Section 210 of the Energy Reorganization Act of 1974 which required the NRC to "develop a plan providing for the specification and analysis of unresolved safety issues relating to nuclear reactors" and "take such action as may be necessary to implement corrective measures with respect to such issues." The ACRS has made significant contributions toward resolution of many generic safety issues (GSIs). One recent example is the Committee's role in the resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance." ACRS was first to express concerns about the effects of chemical reaction products and particle/fiber mats that could form on screens. The Committee was also the first to recognize that increasing screen area, though it could reduce head loss, might result in more fiber debris passing through the screens and increase downstream effects [Nourbakhsh & Banerjee, 2011].

The ACRS was at the forefront of the development of quantitative safety goals. In its May 16, 1979, letter on quantitative safety goals [ACRS, 1979], the ACRS recognized the difficulties and uncertainties in the quantification of risk and acknowledged that in many situations engineering judgment would be the only or the primary basis for a decision. Nevertheless, the Committee believed that the existence of quantitative safety goals and criteria could provide important yardsticks for such judgment. The first set of trial goals (NUREG-0739) [ACRS, 1980] was developed by the ACRS in 1980. These safety goals were the basis for later NRC work on the development of an NRC Safety Goal Policy [NRC, 1986].

Some of the most significant successes of the Nuclear Regulatory Commission were achieved in large part with the benefit of the wise counsel – perhaps even the prodding – of the ACRS"

Excerpt from a speech by Richard Meserve, former NRC Chairman, before a symposium honoring the 50th anniversary of the ACRS, March 4, 2003

In the early 1990s, the ACRS became concerned about the inconsistent use of PRA in the NRC. In a July 19, 1991, letter on the consistent use of PRA [ACRS, 1991], the ACRS acknowledged, "PRA can be a valuable tool for judging the quality of regulation, and for helping to ensure the optimal use of regulatory and industry resources." The Committee also stated that it "would have liked to see a deeper and more deliberate integration of the methodology into the NRC activities." The ACRS also pointed to issues such as the inconsistent use of conservatism and the lack of the treatment of uncertainties. In response to the ACRS, NRC chartered a PRA Working Group and a Regulatory Review Group to review processes, programs, and practices to identify the feasibility of substituting performance-based requirements and guidance founded on risk insights in place of prescriptive requirements [NRC, 2006]. These efforts led the Commission to issue a policy statement on the use of PRA so that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency [NRC, 1995].

The ACRS has been very supportive of the evolution toward a risk-informed and performance-based regulatory system and has taken a leading role in considering some of the challenging issues that have arisen in this effort.

Since early 1990s, the ACRS has played an important role in design certification processes and has identified many technical safety issues during its reviews of advanced LWR designs, which were resolved before the Committee provided its final recommendations for approvals. More discussions on the ACRS's role in design certification review process is provided in Section 3 of this report.

Throughout its history, an essential activity of the ACRS has also been reviewing the research sponsored by the agency. This includes evaluation of technical and programmatic aspects of the overall reactor safety research program as well as episodic review of particularly important ongoing research.

In 1977, Section 29 of the Atomic Energy Act was amended to add the following two sentences: "In addition to its other duties under this section, the Committee, making use of all available sources, shall undertake a study of reactor safety research and prepare and submit annually to the Congress a report containing the results of such study. The first such report shall be submitted to the Congress no later than December 31, 1977." ACRS submitted an annual report on NRC Safety Research Program to Congress from 1977 until 1997. In 1998, Public Law 105-362 struck those two sentences in Section 29.

In 1997, the Commission transferred the research advisory function of the Nuclear Safety Research Review Committee (NSRRC) to the ACRS. In this role, the ACRS was directed to "examine the need, scope, and balance of the reactor safety research program" [NRC, 1997]. The Committee was also directed to "consider how well the Office of Research anticipates research needs and how it is positioned for the changing environment" [NRC, 1997]. Since 1998, ACRS has been submitting reports to the Commission on review and evaluation of the NRC Safety Research Program, annually at first, and after 2004 biennially.

Before the establishment of the Advisory Committee on Nuclear Waste (ACNW) in 1988, ACRS reviewed matters related to the long-term management of radioactive wastes produced within the nuclear industry. In 2007, ACNW was renamed to Advisory Committee on Nuclear Waste and Materials (ACNW&M), and in 2008 ACNW&M was merged into the ACRS. Since then, the Committee has reviewed many aspects of nuclear waste management such as handling, processing, transportation, and storage of nuclear wastes including spent fuel and nuclear wastes mixed with other hazardous substances.

3. ACRS REACTOR LICENSING REVIEWS, A HISTORICAL PERSPECTIVE

3.1 Early Years of Reactor Licensing Reviews

The passage of the 1954 Atomic Energy Act made it possible for private companies to build and operate nuclear reactors under license. This Act also assigned to the AEC the responsibility of protecting the health and safety of the public through a licensing process.

In the early years of development of nuclear power plants, both the technology and its governing regulations were in the formative stages. The early licensing reviews were highly customized and were generally based on the engineering experience and judgment of regulatory staff working closely with the ACRS, without the availability of the regulatory guidance and structure established later during commercial development of LWRs. The AEC safety philosophy, as summarized in a March 14, 1956, AEC letter to the Congress of the United States (responding to a February 16, 1956, letter to the ACRS) [AEC, 1956] was based on the proposition that the ultimate safety of the public depends on three factors:

> *"1. Recognizing all possible accidents that could release unsafe amounts of radioactive materials*
>
> *2. Designing and operating the reactor in such a way that the probability of such accidents is reduced to an acceptable minimum*
>
> *3. By the appropriate combination of containment and isolation, protecting the public from the consequences of such an accident, should it occur."*

However, at the time, the operating experience with power reactors and the state of knowledge of safety analysis had not progressed to the point where it was possible to use quantitative techniques to estimate the probabilities and consequences of accidents. Instead, conservative assumptions were used to provide upper bounds of the potential public consequences resulting from certain hypothetical accidents (the so-called "deterministic" approach). The fundamental concept of defense-in-depth was invoked at the time to ensure that the unquantified probabilities of accidents were small [Nourbakhsh, et.al., 2018].

The Atomic Energy Act of 1954 prescribed a two-step licensing process. The AEC would issue a construction permit based on the safety of the preliminary plant design and the suitability of the prospective site. Only when AEC determined that the plant fully met safety requirements, the applicant would receive a license to load fuel and begin operation. This two-step licensing process allowed enough time to investigate outstanding safety questions and to prescribe modifications to initial plans. It was recognized that the wisdom of permitting construction to proceed without first resolving all potential safety issues was disputable, but there were no alternatives in light of the existing state of the technology and the commitment to the rapid development of atomic power [Walker & Wellock, 2010].

Table 1 depicts the power reactors approved for construction up through 1960. These early power reactors included small prototypes developed by the AEC in cooperation with electric utilities and reactor manufacturers.

Table 1 Early U.S. Nuclear Power Reactors

Name	Type	Power (MWt)	Construction Approved	Operation
Shippingport	PWR	231	1954	1957-1982
Indian Point 1	PWR	585	1955	1962-1974
Dresden 1	BWR	630	1955	1969-1978
Fermi 1	Fast Reactor	300	1955	1963-1972
Yankee	PWR	485	1957	1960-1992
Elk River	BWR	58	1958	1964-1968
Piqua	Organic	48	1959	1963-1966
Carolina-VA	PHWR	63	1959	1963-1967
Hallam	Sodium Graphite	240	1959	1963-1964
Saxton	PWR	20	1959	1967-1972
Pathfinder	BWR	203	1959	1966-1967
Big Rock	BWR	240	1960	1962-1997
Humboldt Bay	BWR	202	1960	1963-1976
Bonus	BWR	50	1960	1967-1974
Peach Bottom 1	HTGR	115	1960	1967-1974

The ACRS concerns during the early years of reactor licensing included reactor siting, need for a dependable containment system as a means of protecting the public against the consequences of reactor accidents, and lack of sufficient information regarding certain features to arrive at a conclusion concerning the construction of the plants.

The Piqua Nuclear Power Facility, an organic cooled and moderated nuclear reactor started operation in 1963 as a demonstration project by the AEC. In 1966, problems with control rods and fouling on cooling surfaces led to ceased operations. The neutron flux within the reactor core induced polymerization of Terphenyl, leading to increased viscosity of the coolant and fouling.

The licensing review of the proposed construction of the 48 MWt Piqua organically cooled and moderated reactor is particularly noteworthy. The reactor designer, Atomic International, did not initially propose a containment based on the argument that no accident had been found which could release significant quantities of radioactive materials. The AEC concluded that containment was needed in a moderately populated region. Atomic International then proposed a new site and an unconventional form of containment. The ACRS wrote a report stating that "*the Committee does not consider the installation at this site of a nuclear power plant of this capacity of a relatively untried type to be without undue public hazard until the proposed unconventional type of containment is replaced by a more substantial and dependable system*" [Okrent, 1981]. Finally, Atomic International proposed a more conventional containment for the Piqua reactor, and the Committee, in its May 18, 1959, report [ACRS,1959], was favorable to the new site.

3.2 Licensing Reviews of Early High-Power LWRs

In the early 1960s, the AEC began defining a standard regulatory prescription to licensing of nuclear reactors. Reactor siting was the first issue addressed with the new approach. Regulations for site selection were developed as 10 CFR Part 100, "Reactor Site Criteria." Part 100 was developed, in part, based on the assumptions that an upper limit of fission product release could be estimated and the containment building, as a final element of defense against the release of radiation, would hold even if a severe accident were to occur. In conjunction with Part 100, the concept of a maximum credible accident was developed to evaluate the acceptability of a potential site (siting limits) and containment design requirements.

After publication of the proposed Part 100 Reactor Site Criteria in 1961, several high-power reactors were proposed for construction. These large LWRs were proposed at sites that did not meet Part 100 without taking credit in calculating off-site doses, either for a reduction in leak rate (e.g., by use of double containment), or by reducing the postulated fission product source available to leak out of containment (e.g., using containment atmosphere cleanup systems, such as containment sprays or closed-loop filter systems) [Okrent, 1979]. The early large LWRs approved for construction using the Part 100 Reactor Site Criteria included San Onofre (1347 MWt PWR), Connecticut Yankee (1473 MWt PWR), and Oyster Creek (1600 MWt BWR). It should be noted that not all reactor proposals received approval and not all approved reactors were ultimately constructed.

The principal focus of the ACRS licensing reviews of early large LWRs appears to have been on the efficacy of the engineered safeguards (containment plus sprays and/or filters) needed to meet the dose guidelines of Part 100. The ACRS licensing reviews also led to evolution of many new safety requirements dealing with a wide range of technical issues. The following are some examples of the issues raised by the ACRS during its licensing reviews of early high-power reactors.

- ***<u>Control Rod Ejection Accidents</u>***: The ACRS report on licensing review of the Connecticut Yankee plant [ACRS, 1964a] was the first to call out the requirements for study of the control rod ejection accident. This led to design changes in large LWRs, either to limit the reactivity worth of control rods or to add an additional mechanical restraint to control rod ejection (an approach taken in BWRs) [Okrent, 1981].

- ***<u>Design Considerations for a Tsunami Following a Major Earthquake</u>*****:** The ACRS report on the proposed 1473 MWt Malibu Nuclear Plant Unit 1 for construction at Corral Canyon (twenty-nine miles west of Los Angles) was the first to raise the issue of the adequate protection against a tsunami following a major seismic event. The following paragraph from the July 15, 1964, ACRS report on the proposed Malibu Plant is particularly noteworthy:

  "*The ability of the plant to withstand the effects of a tsunami following a major earthquake has been discussed with the applicant. There has not been agreement among consultants about the height of water to be expected should a tsunami occur in this area. The Committee is not prepared to resolve the conflicting opinions and suggests that intensive efforts be made to establish rational and consistent*

parameters for this phenomenon. The applicant has stated that the containment structure will not be impaired by inundation to a height of fifty feet above mean sea level. The integrity of emergency in-house power supplies should also be assured by location at a suitable height and by using water-proof techniques for the vital power system. The emergency power system should be sized to allow simultaneous operation of the containment building spray system and the recirculation and cooling system. Ability to remove shutdown core heat under conditions of total loss of normal electrical supply should be assured. If these provisions are made, the Committee believes that the plant will be adequately protected" [ACRS, 1964b].

The proposed Malibu reactor was never built. An intervener group successfully contested the construction of the proposed Malibu plant. The adequacy of seismic design was one of the main points of contention [Okrent, 1981].

- ***Effectiveness of ECCS Design*:** By the mid-1960s, as proposed plants increased significantly in power level, the ACRS became concerned that a core meltdown accident, particularly one in which the plant's emergency core cooling system (ECCS) might fail to operate as designed, could lead to a breach of containment. The ACRS emphasized the need for improved ECCS. By August 1966, General Electric responded in support of the Dresden 3 plant by proposing a redundant core-flooding system and an automatic depressurization system, which would reduce the primary system pressure sufficiently to maximize the effectiveness of the low-pressure core spray or core-flooding system. Later that year Westinghouse introduced accumulators.

- ***Emergency Planning*:** As the size of proposed nuclear power plants increased and containment could no longer be regarded as an unchallengeable barrier to the escape of radioactivity, ACRS paid more attention to emergency planning. In 1966, the Committee noted that many applicants and licensees would rely heavily on local authorities to carry out evacuation if it should become necessary. There were also no guidelines for judging when an evacuation would be advisable. The ACRS decided that it should alert the AEC "*to a problem area where little efforts being exerted*" [Walker, 1992]. Pressed by the ACRS, the AEC undertook a study of emergency plans and procedures that eventually led to adding a new Appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities."

- ***Reactor Pressure Vessel Integrity*:** In 1965, Dresden 2 (a 2250 MWt BWR) was proposed for construction at the site of Dresden Unit 1. Dresden 2 represented a large jump in power level as compared to Oyster Creek (1600 MWt), the largest reactor previously approved for construction. Many ACRS members considered Dresden 2 as a likely prototype for other reactors which would be proposed for metropolitan areas. For this and other reasons, Dresden 2 received much attention by the ACRS, and the potential resolution of certain generic matters such as reactor pressure vessel integrity became tied to its licensing review [Okrent, 1981].

The concern of ACRS for reactor pressure vessel integrity went back to 1961 when the Committee, in its May 20, 1961, report, raised the issue of "*the potential damage to reactor pressure vessel by virtue of the neutron flux to which they are subjected during their life*" [ACRS, 1961]. However, prior to Dresden 2 review, the vessel failure was not considered "credible".

Dresden Nuclear Power Station, Unit 2 (a 2250 MWt BWR 4 with Mark 1 Containment}, proposed for construction in 1965, represented a large jump in power level as compared to the largest reactor previously approved for construction. For this and other reasons, Dresden 2 received much attention by the ACRS, and the potential resolution of certain generic matters tied to its licensing review.

In 1965, the concern about pressure vessels had been growing, in part, due to the 1964 failure of a very large heat exchanger at a temperature near the nil ductility temperature while being tested by the Foster Wheeler Corporation [Okrent, 1981]. During the Dresden 2 licensing review, the ACRS discussed this matter extensively and debated whether this issue should be handled in a generic way or by specifically addressing this applicant. The ACRS finally decided to issue a report favorable to the construction of Dresden 2 and at the same time wrote a general report regarding reactor pressure vessels.

- ***Anticipated Transients Without Scram (ATWS)*: ** The issue of ATWS was first raised by E. P. Epler, an ACRS consultant, in a January 21, 1969, letter to the ACRS executive secretary. Soon after, the ACRS decided to identify the issue in its letter reports on Hatch unit 1 [ACRS, 1969a] and on the application for the construction authorization for the Brunswick Units 1 and 2 {ACRS, 1969b]. In each report the Committee recommended "*a study be made by the applicant of further means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram during anticipated transients.*"

3.3 Reviews of Latter Non-LWR Designs

The ACRS has a long history of review and evaluation of non-LWRs. Early safety reviews of non-water-cooled power reactors (refer to Section 3.1), like those performed for the early LWRs, were highly customized and were generally based on the engineering experience and judgment of the regulatory staff working closely with the ACRS, without the availability of the regulatory guidance and structure established later during LWR commercial development. In latter reviews, explicit use was made of LWR regulatory guidance where applicable.

Prior to the 1986 policy statement on regulation of advanced reactors [NRC, 1986], the principal statement on non-LWR review policy was given in the introduction to Appendix A to 10 CFR 50: General Design Criteria (GDC) for Nuclear Power Plants. Specifically, this introduction states: "*These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power*

plants similar in design and location to plants for which construction permits have been issued by the Commission. ***The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.*** "

The regulatory staff worked closely with the ACRS in developing the GDC. Versions of general design criteria were publicly available as early as 1965 before initially being incorporated into Part 50 in February 1971.

Development of the GDC led to the "comparable level of a safety" philosophy under which proposed high temperature gas-cooled reactors (HTGRs) and liquid metal reactors (LMRs) were reviewed in later years [NRC,1988]. This philosophy was based on the notion that a comparable level of safety would be established for all reactor types, recognizing that the licensing criteria for non-LWRs could be developed, to the extent feasible, using those for LWRs. This philosophy was implemented, with respect to the existing criteria, by direct adoption, suitable adaptation, or development of design-specific criteria (if needed). Direct adoption of the existing criteria in many instances could provide a ready means of ensuring a comparable level of safety [NRC, 1988].

For those existing criteria that could not be viewed as clearly applicable, suitable adaptations were developed to permit the use of the phrase "meets the objectives of" or words to this effect. Development of such adaptations was usually a straightforward practice of the applicant identifying and justifying discrepancies from the criteria [NRC,1988]. An early example of the adaptive approach was the means for conformance of the Fort St. Vrain design to the GDC for LWRs.

For those portions of non-LWR designs that were uniquely different from those of LWR designs (e.g., requirements for handling a sodium coolant or the use of a concrete reactor vessel for HTGRs), adoption or adaptation of existing regulations or standards was not possible or desirable. In such cases, design-specific licensing criteria were developed by engineering judgment and analysis.

Fort St. Vrain Generating Station: A 842 MWt HTGR cooled with helium. The prestressed concrete reactor vessel (PCRV) of this plant was the first in the United States. The ACRS issued a favorable, but cautious report on its construction permit. It operated sporadically for ten years (1979-1989). …

Table 2 depicts ACRS review activities associated with latter non-LWR designs. Short summaries of those reviews are provided in the following:

Fort St. Vrain: The ACRS licensing review of the proposed construction of Fort St. Vrain is particularly noteworthy. Fort St. Vrain was a HTGR cooled with helium and designed to produce 842 MW_t (330 MW_e). The prestressed concrete reactor vessel (PCRV) proposed for this plant was the first in the United States. This PCRV was to contain not only the core, but the entire primary coolant system. The plant utilized a confinement building equipped with ventilation filters for removing particulates and iodine from the building exhaust.

Table 2 The ACRS Review of Latter Non-LWR Designs

Design	ACRS Reviews	Remarks
Fort St. Vrain – HTGR, 842 MWt	ACRS Report on CP May 15, 1968 Final ACRS Report, May 12, 1971	ACRS issued a favorable, but cautious report on CP. Sporadic operation (1979-1989), mainly caused by water ingress from helium circulator bearings.
1100 MWe HTGR (A 1969 study to upgrade HTGR power level)	ACRS reviewed the conceptual design ACRS Report: November 12, 1969	A conventional low-leakage containment building similar to those used for PWRs determined to be necessary for an HTGR of this size.
Gas Cooled Fast Breeder Reactor, GCFBR	Concept was reviewed by ACRS, 1971-1974 ACRS Report, November 8, 1974	ACRS indicated certain safety disadvantages unique to the GCFBR, as well as some safety problems common to all fast reactors. General Atomic pursued a program to resolve the outstanding safety/licensing issues.
Summit and Fulton Plants HTGR, 700-1000 MWe.	Licensing activities 1973 to 1975 ACRS Reports issued March 12, 1975 April 8, 1975	Sited in Delaware and Pennsylvania, but plants cancelled for economic reasons prior to public hearings and CP issuance.
FFTF (Fast Flux Test Facility) sodium-cooled fast reactor, 400 MWt	Safety review by the ACRS, but a license was not required by law ACRS Final Report November 8, 1978	Operated (1982-1992) as a national research facility to test various aspects of commercial reactor design and operation.
Clinch River Breeder Reactor, CRBR liquid-sodium-cooled fast-breeder reactor. 975 MWt	ACRS reviewed the CP application Final Report: April 19, 1983	NRC completed its review and public hearing for CP in 1983. Plant never built due to termination of the project by Congress.

On May 15, 1968, the ACRS issued a favorable, but cautious report on its review of the Fort St. Vrain CP application, with supplemental added comments by two members [ACRS, 1968b]. In its report, the ACRS stated that since this was a first-of-a-kind reactor, particular attention must be paid to final design, construction, and quality control. The ACRS report also identified several items of safety-oriented research and development as critical to the safety of this system. Fort St. Vrain operated sporadically for a little more than 10 years (1979-1989). In 1989, during a plant shut down to repair a stuck control rod pair, numerous cracks were discovered in several steam generator main steam ring headers. The required repairs were determined to be too extensive to justify continued operation.

1100 MWe HTGR: In the late 1960s, the ACRS reviewed a conceptual design and proposed design bases for an 1100 MWe HTGR. The reactor concept utilized a helium-cooled, graphite-moderated ceramic fuel core similar to that of the Fort St. Vrain reactor. In its November 12, 1969, report, the ACRS

stated that although insufficient information has been provided for the Committee to make a recommendation as to the adequacy of the specific design bases proposed, it believes that an 1100 MWe HTGR plant along the general lines of the preliminary concept described in the referenced documents may constitute a system which can be so engineered and operated [ACRS, 1969c].

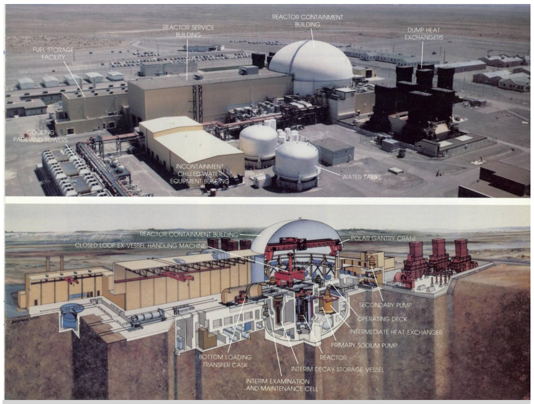
Summit and Fulton Plants: In the 1970s, the ACRS also reviewed the Summit and Fulton applications for large HTGRs that were not subsequently built.

The proposed Summit Power Station consisted of two nuclear units, each using a General Atomic HTGR having a rated power level of 1532 MWe. The Summit Power Station would have been located in Delaware, approximately 15 miles from Wilmington. In its March 12, 1975, report, the ACRS recognized that the Summit Power Station represented a new design so that many of the proposed systems and components were relatively untested at the time. This aspect was apparent in the NRC Staff Safety Evaluation Report (SER) where several items were unresolved, or resolution was to be deferred until the post-construction permit period. The Committee urged the resolution of those outstanding items well before equipment was installed [ACRS, 1975a].

The proposed Fulton Generating Station consisted of two nuclear units, each using a General Atomic HTGR having a rated power level of 1160 HW(e). The Fulton Generating Station would have been located approximately 17 miles south of Lancaster, Pennsylvania. The similarities between the Fulton and Summit Stations were considered in the ACRS review of Fulton [1975b]. With no significant exceptions, the Committee's concerns were generic to both the Fulton and Summit Stations and to all large HTGRs. The ACRS comments in its April 8,1975, report on Fulton [ACRS, 1975b] were essentially the same as those made on the Summit Power Station.

GCFBR: In the mid-1970s, the ACRS reviewed a conceptual design of a gas cooled fast breeder reactor (GCFBR) by General Atomic Company. The purpose of this review was to acquaint the Committee with the conceptual design and to identify those areas which the Committee believed required further technological development, or which the Committee deemed unacceptable at the time. In its November 8,1974, report [ACRS, 1974], the ACRS indicated certain safety disadvantages unique to the GCFBR, as well as some safety problems common to all fast reactors. Based on the licensing concerns of the ACRS and NRC, General Atomic pursued a program to resolve the outstanding safety/licensing issues. One example was studies on how other fast and thermal reactor systems could provide a combination of built-in thermal capacity and/or natural circulation cooling, avoiding the need for rapid restoration of forced circulation of primary coolant to prevent fuel overheating. Studies were also performed on other GCFR design arrangements which could provide sufficient natural circulation cooling if coolant pressure were maintained [Lipinski et. al, 1978].

FFTF: In 1970s, the ACRS reviewed the proposed construction and operation of the DOE Fast Flux Test Facility (FFTF) [ACRS,1978]. The FFTF was a 400 MWt sodium-cooled fast reactor located at DOE's Hanford Reservation. Since the FFTF was a DOE facility, the scope of the review was defined by the DOE request that NRC provide only advice regarding the adequacy of its design and technical specifications to ensure safe operation.



Fast Flux Test Facility (FFTF): A 400 MWt sodium-cooled fast reactor located at the DOE's Hanford Reservation. The ACRS provided advice regarding the adequacy of its design and technical specifications. The FFTF operated (1982-1992) as a national research facility to test various aspects of commercial reactor design and operation.

The FFTF did not have a Class 1E power supply to provide decay heat removal. Instead, the project depended upon natural convection cooling in the event of loss of offsite power and failure of the onsite diesel generators. The project's calculations indicated that natural circulation would provide decay heat removal. It was proposed that the natural circulation decay heat removal be measured during the startup testing. In its November 8, 1978, report, the ACRS concurred that the adequacy of the decay heat removal by natural circulation should be experimentally verified [ACRS, 1978]. The FFTF operated (1982-1992) as a national research facility to test various aspects of commercial reactor design and operation.

CRBRP: In the early 1980s, the ACRS reviewed the CP application for the Clinch River Breeder Reactor Plant (CRBRP). The CRBRP was to be a liquid-sodium-cooled, mixed-oxide-fueled, fast-breeder reactor demonstration power plant. Design power was 975 MWt (350 MWe).

The applicants were conducting a full-scope PRA on the CRBRP design during its licensing review by the NRC. In its April 19, 1983, report, the ACRS recommended that timely completion of the PRA by the applicants, to permit its review and evaluation by the NRC Staff and the ACRS, be a condition of the construction permit. ACRS also raised a number of safety issues for which more work must be done prior to their resolution. The ACRS believed that, if the matters raised by the Committee and the open items described in the SER were resolved in a satisfactory manner, the CRBRP could be constructed with reasonable assurance that it could be operated without undue risk to the health and safety of the public [ACRS, 1983].

The NRC completed its review and public hearing for CP in 1983. The plant was never built due to termination of the project by Congress.

3.4 Pre-application Reviews of Earlier Advanced Non-LWR Designs

In 1986, the Commission issued its policy statement on advanced reactors [NRC, 1986]. Advanced reactors were to include evolutionary LWRs, non-LWRs, and small modular light water reactors. In its policy statement, the Commission expected that advanced reactors would provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions. The Commission's policy statement regarding regulation of advanced nuclear power plants has encouraged early interaction (prior to a license application) between vendors and the NRC "*to provide for early identification of regulatory requirements for advanced reactors, and to provide all interested parties, including the public, with a timely, independent assessment of the safety characteristics of advanced reactor designs.*" The NRC has been particularly interested in any regulatory issues which could lead to the need for Commission policy decisions, or technical issues

unique to the design that could require extensive effort and a long lead time to resolve.

Consistent with the policy statement, the ACRS has also been holding pre-application meetings and discussions to familiarize the Committee with the design and to identify topics for more detailed discussions before the application is submitted. Table 3 depicts the ACRS pre-application reviews of non-LWR designs in the late 1980s and early 1990s. Short summaries of those pre-application reviews are provided in the following.

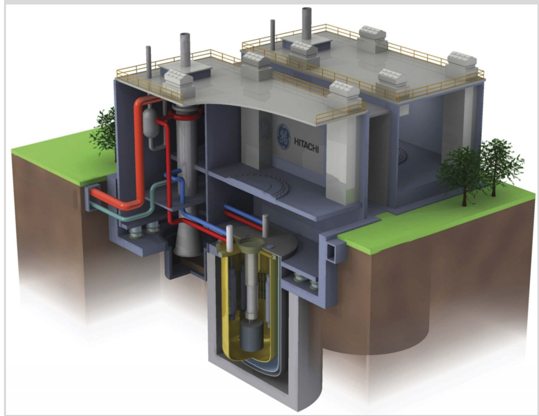
MHTGR: The modular high temperature gas cooled reactor (MHTGR) concept was a product of a joint DOE/industry program to develop a design for a nuclear power plant using HTGR technology and having important inherently safe characteristics. In its October 13, 1988, report [ACRS, 1988] on preapplication safety evaluation of MHTGRs, the ACRS raised a number of safety issues to be adequately addressed to assure the key safety characteristics claimed for the design would be realized in an actual plant.

Table 3 The ACRS Pre-application Reviews of Non-LWR Designs

Design	ACRS Reviews	Remarks
MHTGR (Modular HTGR) 350 MWt, annular prismatic block core arrangement	Preapplication design review by ACRS ACRS report: October 13,1988	The ACRS raised a number of safety issues that had to be adequately addressed to assure the key safety characteristics claimed for the design would be realized in an actual plant.
SAFR (Sodium Advanced Fast Reactor) 3600 MWt (each "power pack" comprising four reactor modules)	Preapplication design review, ACRS Report: January 19, 1989	The conceptual design was under review when the DOE terminated the work in September 1988.
PRISM (Power Reactor Innovative Small Module) Pool-type, LMR. 475 MWt	Preapplication Design Review, 1986-1994 Final ACRS Report: November 10, 1993	The staff's preliminary findings were reviewed by the ACRS. After the review, several revisions to the conceptual design were made.
Toshiba 4S (Super-Safe, Small, and Simple) 30 MWt pool type LMR	Preapplication review meetings in 2007-2008 ACRS was briefed and held discussions regarding the design	The NRC ceased its review of the Toshiba 4S in 2013 without issuing any review documents.

SAFR: The Sodium Advanced Fast Reactor (SAFR) conceptual design was another product of a DOE program to develop designs for possible future power reactor systems that would have enhanced safety characteristics. Other design projects in the program were the MHTGR and the PRISM. The SAFR conceptual design was under review when the DOE decided to discontinue its development and concentrate LMR efforts in the PRISM design organization. In its January 19, 1989, report on safety evaluation of SAFR design, the ACRS stated that a continuing program of research and development, including plans for extensive prototype testing, would be necessary to support further design [ACRS, 1989]. The Committee also commented on a number of specific safety issues that DOE should consider if it continued design and development of the SAFR concept.

PRISM: The PRISM was the only DOE sponsored design that was developed to the point that a safety review was conducted. The NRC conducted a thorough review of the 475 MWt design between 1986 and 1994. Consistent with the Commission's advanced reactor policy, the staff, to the extent feasible, used existing regulations at the time to formulate criteria and procedures for review of this design. The ACRS reviewed the staff's preliminary findings. After the review, several revisions to the conceptual design were made. Because the staff review was based on a conceptual design, the preapplication safety evaluation report (PSER) did not, nor was it intended to, result in an approval of the design. Instead, it identified certain key safety issues, provided some guidance on applicable licensing criteria, assessed the adequacy of the research and development programs, and concluded that no obvious impediments to licensing the PRISM design had been identified. In an ACRS report, dated November 10, 1993, on the draft PSER for the PRISM liquid-metal reactor [ACRS,1993], the Committee stated that "*although our own review of the PSER was less detailed than would have been appropriate for a safety evaluation report on an actual application, we believe that the staff has satisfactorily fulfilled its role in the preapplication process.*"



Power Reactor Innovative Small Module (PRISM): The NRC conducted a thorough review of the 475 MWt design between 1986 and 1994. Consistent with the Commission's advanced reactor policy, the NRC staff, to the extent feasible, used existing regulations at the time to formulate criteria and procedures for review of this design. The ACRS reviewed the staff's preliminary findings. After the review, several revisions to the conceptual design were made.

Toshiba 4S: The Toshiba 4S (Super-Safe, Small, and Simple) was a small sized (30 MWt) sodium-cooled fast reactor designed for remote locations with small grids. The NRC held a series of preapplication review meetings in 2007-2008. From 2008 to 2013, Toshiba continued to submit a series of technical reports that responded to the policy statement on the licensing of advanced reactors. In 2010, the ACRS was briefed by and held discussions with representatives of the NRC staff regarding the Toshiba 4S design. Toshiba did not proceed with an application for certification of the design. NRC ceased its review of the Toshiba 4S in 2013 without issuing any review documents [NRC, 2019].

3.5 Licensing Reviews of Advanced Light Water Reactors

In the early 1980s, in cooperation with DOE, the U.S. nuclear utility industry, with support from the Electric Power Research Institute (EPRI), initiated the Advanced Light Water Reactor Program to ensure a viable nuclear power generation option for the 1990s and beyond. A major objective of the program was to develop designs for future LWRs that were safer, more reliable, easier to operate, and more certain of being licensed without delays. A means to assure this outcome was the development of the ALWR Utility Requirements Document by senior, experienced utility personnel in the U.S. and overseas that incorporated the lessons learned from decades of worldwide operating experience with LWRs [DOE, 2001]. The ACRS followed the development of the EPRI ALWR program from its inception and offered suggestions regarding safety improvements on several occasions [ACRS, 1992].

In 1989, the NRC established alternative licensing processes to improve regulatory efficiency and add greater predictability to the licensing process. The alternative licensing processes include a combined license that essentially combines a construction permit and an operating license, with certain conditions, into a single license. Other licensing alternatives established in 1989 are early site permits, which allow an applicant to obtain approval for a reactor site for future use, and certified standard plant designs, which can be used as pre-approved designs. The NRC may approve and certify a standard nuclear plant design through a rulemaking, independent of a specific site and an application to construct or operate a plant. A design certification is valid for 15 years from the date of issuance but can be renewed for an additional 10 to 15 years. According to NRC regulation (10 CFR 52.53), the design certification application (DCA) is referred to the ACRS for a review and report.

Advanced Passive 1000 (AP1000): A two-loop pressurized water reactor (PWR) with passive safety features. The ACRS played an important role in the AP1000 design certification process by providing an independent review of the NRC staff's determination of compliance with the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations. The ACRS identified many technical issues during its review process which were resolved before the Committee provided its final recommendations on the design certification.

The ACRS has played an important role in design certification processes by providing an independent review of the determination of compliance with the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations.

The ACRS has identified many technical issues during its design certification reviews, which were resolved before the Committee provided its final recommendations for approvals. Table 4 lists the design certification applications that the ACRS has reviewed to date. Short summaries of those design certification reviews are provided in the following.

Table 4 The Design Certification Applications Reviewed by the ACRS

Design	Applicant	Final ACRS Report
Advanced Boiling Water Reactor (ABWR)	General Electric (GE) Nuclear Energy	April 14, 1994
System 80+	ASEA Brown Boveri - Combustion Engineering (ABB-CE)	May 11, 1994
Advanced Passive 600 (AP600)	Westinghouse Electric Company	July 23, 1998
Advanced Passive 1000 (AP1000)	Westinghouse Electric Company	July 20, 2004
Economic Simplified Boiling-Water Reactor (ESBWR)	GE Hitachi Nuclear Energy	October 20, 2010 April 17, 2014
U.S. EPR	AREVA NP, Inc.	Suspended
U.S. Advanced Pressurized-Water Reactor (US-APWR)	Mitsubishi Heavy Industries, Ltd	Suspended
Advanced Power Reactor 1400 (APR1400)	Korea Electric Power Corporation and Korea Hydro & Nuclear Power Co., Ltd.	July 26, 2018
NuScale Small Modular Reactor	NuScale Power, LLC	July 29, 2020

ABWR: The Advanced Boiling Water Reactor (ABWR) is a forced circulation boiling water reactor with a rated power of 3926 MWt. The U.S. version of the ABWR standard design utilized a significant portion of the detailed design information developed jointly by General Electric Nuclear Energy (GENE), Hitachi, and Toshiba for the international version which was being built in Japan.

The ACRS reviewed the ABWR design and issued its final report on the safety aspects of the GENE application for certification of the ABWR design in April 1994 [ACRS,1994a]. The ACRS played an important role in GENE ABWR design certification process [Nourbakhsh & Banerjee, 2015]. The ACRS review of ABWR design led to significant design changes, such as a GENE proposal that safety-related equipment inside of the ABWR secondary containment be environmentally qualified for steam at 15 psig and about 248°F and adding a third break isolation valve in the 8-inch Reactor Water Cleanup (CUW) supply line located inside of primary containment.

In its April 14,1994, report, the ACRS stated that acceptable bases and requirements had been established in the application to assure that the U.S. version of the ABWR standard design could be used to engineer and construct plants that with reasonable assurance could be operated without undue risk to the health and safety of the public {ACRS, 1994a].

System 80+: The ASEA Brown Boveri - Combustion Engineering (ABB-CE) System 80+ standard plant design evolved from the CE System 80 plant design. The ABB-CE System 80+ design included several features that would have enhanced safety relative to past PWR designs. Some of those features resulted from the use of PRA methodology by ABB-CE during the System 80+ design process.

Based on the results of ACRS review of those portions of the ABB-CE System 80+ application which concerned safety, the Committee concluded that the System 80+ standard plant design can be used to engineer and construct plants that, with reasonable assurance, can be operated without undue risk to the health and safety of the public [ACRS, 1994b].

AP600: The 1933 MWt (600 MWe) Advanced Passive 600 (AP600) design represented a significant departure from previous commercial nuclear reactor technology in that it placed more dependence on passive systems for accident response. Unique features of the AP600 design included an improved reactor core design, a large reactor vessel, a large pressurizer, an in-containment refueling water storage tank (IRWST), an automatic depressurization system, a digital microprocessor-based I&C system, hermetically sealed canned rotor coolant pumps mounted to the steam generator, and increased battery capacity.

Westinghouse conducted an extensive test and analysis program, utilizing separate-effects and integral-system facilities both to investigate the behavior of the AP600 passive safety systems and to develop a database for validation of the computer codes used to perform accident and transient analyses. During the extensive reviews of the Westinghouse test and analysis program, the ACRS raised numerous safety issues, which were adequately resolved before the Committee concluded that "the AP600 design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public" [ACRS, 1998].

AP1000: The AP1000 design is similar in concept to the AP600 design but provides much higher power levels (1000 MWe for AP1000 compared to 600 MWe for AP600). The ACRS concluded that most of the AP600 review findings were applicable to the AP1000 design. This conclusion greatly enhanced the efficiency of the reviews of the AP1000 safety assessments [ACRS, 2004].

During ACRS review process, the Committee identified many technical safety issues of concern and areas for which it needed additional discussions [Nourbakhsh, et al., 2013]. The ACRS agreed with the staff's proposed resolution of all but two of those issues. The Committee developed its own arguments for the resolution of the remaining two issues [ACRS, 2004].

ESBWR: The Economic Simplified Boiling-Water Reactor (ESBWR) is a direct-cycle, natural circulation BWR and has passive safety features to cope with a range of design basis accidents (DBAs).

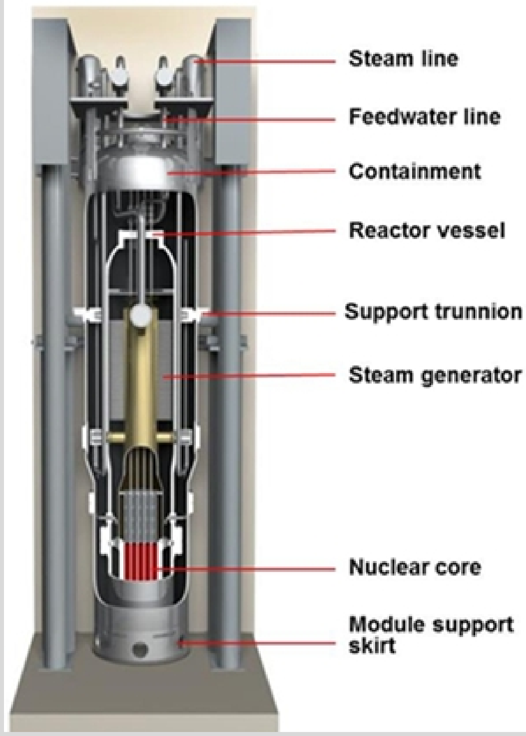
During its review, the ACRS identified many issues of concern and areas for which the Committee needed additional discussion. Those issues included combustion control of flammable non-condensable gases in the passive containment cooling system (PCCS), which led the General Electric-Hitachi Nuclear Energy (GEH) to revise the design of the isolation condensers (IC) and PCCS to address the potential for hydrogen detonations within the condenser tubes or the lower plenum [ACRS, 2010]. GEH also provided more detailed explanations and tabular information in the ESBWR Design Control Document revisions to give the Committee confidence that the four fundamental principles are inherent in the hardware and software DI&C architectures, i.e., redundancy, independence, determinate behavior, and diversity and defense in depth [ACRS, 2010].

In its report, dated October 20, 2010, the ACRS concluded that the ESBWR design is robust and there is reasonable assurance that it can be built and operated without undue risk to the health and safety of the public [ACRS, 2010].

APR1400: The APR1400 is a pressurized-water reactor, which evolved from the System 80+ design. APR1400 includes several features that are designed to further improve safety and operability over that of the System 80+. Enhancements include the combination of four trains of safety injection with direct vessel injection, and a unique safety injection tank fluidic device, which optimizes the safety injection flow rate during the initial blowdown and subsequent, long-term, core reflood phase. The performance of the fluidic device was verified via full scale testing. The plant is designed to be "fiber free," which along with experimental verification using conservative conditions, benefits long-term core cooling following a LOCA.

During its reviews, the ACRS identified many issues of concern and areas needing additional discussion that were addressed in nine ACRS letters. In its report, dated July 26, 2018, the ACRS concluded that "the APR1400 design is mature and robust. There is reasonable assurance that it can be constructed and operated without undue risk to the health and safety of the public" [ACRS, 2018].

NuScale: The certified NuScale design consists of up to 12 NuScale Power Modules (NPMs) in a single reactor building. The NPMs are largely immersed in a large pool of borated water, also serving as the ultimate heat sink. Each NPM is a small, integrated, natural-circulation PWR composed of a shrouded reactor core and riser, a pressurizer, and two helical-tube steam generators within a reactor pressure vessel, and housed integral to a high-strength, steel containment vessel that closely surrounds the reactor vessel. Each NPM is rated at 160 MWt, with an output of approximately 50 MWe.



NuScale Small Modular Reactor: An integral pressurized-water reactor designed by NuScale Power LLC. The ACRS identified many technical safety issues during the NuScale design certification review. The ACRS review led to design and setpoint changes to the NuScale Power Module

The ACRS identified many technical safety issues during the NuScale design certification review. As a result of the ACRS review, NuScale incorporated design and setpoint changes to the NuScale Power Module to mitigate the effects of boron dilution in the downcomer for uncontrolled passive cooling events [ACRS, 2020a].

In its report, dated July 29, 2020, the ACRS concluded that there is reasonable assurance that the NuScale small modular reactor can be constructed and operated without undue risk to the health and safety of the public [ACRS, 2020b].

4. LOOKING AHEAD TO ACRS LICENSING REVIEWS OF NEW ADVANCED REACTOR DESIGNS

4.1 Essential Role of ACRS in Reactor Licensing Reviews

Throughout ACRS's history, the Committee's independent review has been an essential element of the reactor licensing process. The Committee's licensing reviews led to evolution of many new safety requirements and design changes dealing with a wide range of technical issues.

A March 14, 1956, AEC letter to the Congress [AEC, 1956], in response to a letter to ACRS on the question of public safety of nuclear reactors, states the following:

> "*The financial incentive of the owners of the reactor to take all steps necessary to protect their investment, as well as to decrease their potential public liability, and the legal and moral responsibilities of the Commission to protect the public from overexposure to radioactivity, are resulting in a system which is characterized by an attitude of caution and thoroughness of evaluation unique in industrial history. Every phase of the reactor design and operating procedure is reviewed separately and as a part of the whole. The inherent nuclear, chemical, metallurgical, physical, and mechanical characteristics of the fuel, moderator, coolant, neutron absorbers, and structural materials are carefully considered to assure that the probability of an operating mishap has by adequate design and operating precautions been brought to an acceptably low level*."

As the NRC staff strategizes to assure that the Agency is ready to review potential licensing applications for new advanced reactor designs, the role of ACRS, with its diverse technical expertise, will continue to be essential for an independent integrated/multi-disciplinary review. The Committee's role is particularly important because the new advanced reactor designs, currently under development, pose new challenges for safety-licensing reviews. These challenges include:

- First-of-a-kind designs with a variety of coolants, fuel forms, and innovative configurations

- Designs that do not have the same levels of operating and regulatory experience as that of LWRs

- Limited experimental database and validation

- Implementation of a new licensing approach

4.2 Enhancing the Efficiency of the Review Process

The ACRS continues to perform introspective evaluations to identify ways to improve its own effectiveness and efficiency as the NRC accomplishes its mission within a changing regulatory framework and culture, and in an expanding industry environment. Over the past several years, the ACRS has also collaborated with NRC program offices to achieve greater efficiencies while maintaining its independence [ACRS, 2019].

As part of its continuing effort to become more effective and assist the NRC in its transformation initiatives, the ACRS recently conducted a self-assessment. This self-assessment was based on the Committee's observations and lessons-learned from its recent NuScale design certification and standard design approval application reviews, informed also by its prior design certification and early site permit reviews, as well as interactions with the NRC staff. Observations and lessons-learned from the ACRS self-assessment led to several recommendations by the Committee that could improve future NRC reviews of advanced reactor designs [ACRS, 2020c]. For a more effective and expeditious review, the ACRS has adopted a cross-cutting approach, focusing on key safety-significant design issues. It is expected that this will streamline reviews, resulting in more efficiency and shorter schedules.

Ultimately it is the completeness and quality of a license application and associated supporting documents that significantly impacts the efficiency of the review process (both for the NRC staff and the ACRS). The desired attributes that would improve the quality and completeness of future applications of advanced reactor designs include the following:

Completeness of the Design

Design completeness has a profound impact on the efficiency of the review process. Proposed new reactor designs should be sufficiently complete to demonstrate that all Structures, Systems, and Components (SSCs) important-to-safety are appropriately identified, designed, and tested, to be commensurate with their functions and to provide adequate defense in depth.

Without having an "essentially complete design" and performing detailed component and system analysis, it may be difficult for NRC to make a technically sound finding on any requested deviation (exemption) from historical regulatory requirements (e.g., General Design Criteria).

Design changes during the review process, as observed in the past, may also adversely impact the efficiency of the review process.

Comprehensiveness of Knowledge Base

All safety decisions are based, either explicitly or implicitly, on identifying radiological hazards and addressing the "risk triplet" questions: "What can go wrong?"; "How likely is it?"; and "What are the consequences?" The NRC addresses these three questions through the body of its regulations and guidance. The comprehensiveness of the knowledge base (experimental data base, operational experience, relevant analyses, etc.), to support the safety decisions, has significant impacts on the review process efficiency.

Both traditional deterministic and probabilistic approaches to safety analyses are based on identification of hazards, initiating events that disturb normal operation, scenarios (event sequences) that could evolve from the initiating events, and their associated consequences. Theoretical and experimental bases are needed for understanding the associated phenomenology of possible scenarios.

For the new non-LWRs, their design maturity and knowledge base are not likely to be as comprehensive as for evolutionary LWR-based designs. The limited knowledge base may impact the regulatory review.

When there is a lack of operating experience or an inability to perform

experiments with sufficient similitude to the planned full-scale design, one approach for dealing with limited knowledge base, as suggested by the ACRS, is limitations on power ascension and focused surveillance tests during initial operation [ACRS, 2020d].

Proper Consideration of Uncertainties

Safety-licensing decisions are made in the face of uncertainties and within the boundaries of the state of knowledge of how the proposed reactor design would behave under both normal and accident conditions. Both deterministic and probabilistic safety evaluations must deal with uncertainties. Proper consideration of uncertainties significantly helps the review process. Addressing uncertainties affects the reviewer confidence regarding the results of safety evaluations and the resulting safety margins.

Two major groups of uncertainty that have been recognized are aleatory (or stochastic) and epistemic (or state-of-knowledge) uncertainty. The key distinction between these two types of uncertainty is that aleatory uncertainty is irreducible. Epistemic uncertainty, in contrast, can be reduced by further study.

There are two classes of epistemic uncertainty: parameter uncertainty and model uncertainty. Parameter uncertainties are those associated with the values of the fundamental parameters of a model, such as equipment failure rates that are used in quantifying the accident sequence frequencies in PRAs. Model uncertainties reflect the limited ability to model accurately the specific events and phenomena. Completeness, including possible “unknown unknowns,” can also be considered as one aspect of model uncertainty. Completeness uncertainty arises because not all contributors to risk are addressed in PRA models and not all phenomena and processes are addressed in deterministic safety evaluation models. The safety philosophy of defense in depth and safety margins has been the traditional means of dealing with uncertainties.

The novel aspects of new technologies and first-of-a-kind reactor concepts make the identification of hazards, initiating events, and scenarios more challenging. To address uncertainties caused by limited information, the ACRS has recommended critical examination of the design, its safety behavior, and all aspects of operations, starting from a “blank sheet of paper” to avoid bias. The Committee has also suggested use of several analysis tools that have been developed to improve the search process. These tools apply equally to traditional and probabilistic safety analyses [ACRS, 2020c]. Such analysis tools can help formalize and add structure to the safety assessment and reduce completeness uncertainty.

Appropriate Timing of Supporting Documents Submittals

The proper timing of the supporting documents (e.g., licensing topical reports) submittals may also have a significant impact on the efficiency of the review process.

Submittal of critical licensing topical reports late in the review process or parallel with related chapters of the DCA, as observed in the past, could reduce efficiency. The proper timing would be the sequential hierarchical order of submittals wherein licensing topical reports on methodology description, demonstration, and verification and validation precede their applications.

The proper timing of critical topical reports submittals is vital for review of

non-LWR concepts, which are likely to have more uncertainty associated with analytical methods and their application, underlying experimental bases, and validation of models. The licensing topical reports that support the design basis and safety analyses should be reviewed as early in the process as possible because new reactor designs, especially non-LWRs, will generally be more dependent on analytical methods for understanding the safety response of the system [ACRS, 2020c].

5. SUMMARY AND CONCLUSIONS

For almost 65 years the ACRS has had a continuing statutory responsibility for providing independent reviews of, and advising on, the safety of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards in the United States.

Throughout ACRS's history, the Committee's independent review has been an essential element of the reactor licensing process. The Committee's licensing reviews led to evolution of many new safety requirements and design changes dealing with a wide range of technical issues.

As the NRC staff strategizes to assure that the Agency is ready to review potential licensing applications for new advanced reactor designs that are radically different from the current fleet of LWRs, the role of ACRS, with its diverse technical expertise, will continue to be essential for an independent integrated/multi-disciplinary review.

The ACRS continues to perform introspective evaluations to identify ways to improve its effectiveness and efficiency as the NRC accomplishes its mission within a changing regulatory framework and culture, and in an expanding industry environment. Over the past several years, the ACRS has also collaborated with NRC program offices to achieve greater efficiencies while maintaining its independence.

Ultimately it is the completeness and quality of a license application and associated supporting documents that significantly impacts the efficiency of the review process (both for the NRC staff and the ACRS). The desired attributes that would improve the quality and completeness of future applications of advanced reactor designs include design completeness, knowledge base (experimental data base, operational experience, relevant analyses, etc.) comprehensiveness, proper consideration of uncertainties, and appropriate timing of supporting document submittals.

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